BNL-107113-2015-CP

PWR Plant Model to Assess Performance of Accident Tolerant Fuel in Anticipated Transients and Accidents

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Accident Tolerant Fuel Concepts for Light Water Reactors Conference
Oak Ridge National Laboratory
October 13-17, 2014

December 2014

Nuclear Science & Technology Department

Brookhaven National Laboratory

U.S. Department of Energy
Office of Fuel Cycle Technologies

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Abstract. A PWR plant model based on the reactor system code TRACE has been assembled to enable the simulation of a broad spectrum of anticipated operational occurrences (AOO) and design basis accidents (DBA). The objective is to provide a simulation platform to support the development of advanced fuels and claddings and the deployment of accident tolerant fuel (ATF) in current PWR designs. The TRACE model simulates a Westinghouse 2308-MWt three-loop PWR, including standard primary and secondary loop components and various trips and control systems to model plant responses. Kinetics parameters for the standard UO_2 fuel and nitride fuel (UN-U_3Si_2-UB_4) have been developed from PARCS stand-alone full-core calculations to provide inputs to the TRACE point-kinetics model. Several transients have been analyzed to assess the performance of ATF relative to standard Zircaloy-clad UO_2 fuel. These include complete loss of primary flow, steam generator tube rupture, small break (SB) and large break (LB) loss-of-coolant-accidents (LOCA). Among the accidents analyzed, the largest performance difference between the oxide fuel and the nitride fuel is in the LBLOCA where the peak clad temperature (PCT) for the nitride fuel is 30 K lower than that of the oxide fuel.

1. Introduction

A PWR plant model based on the reactor system code TRACE has been assembled to enable the simulation of PWR transients and accidents. The objective is to provide a simulation platform to support the development of advanced fuels and claddings and the deployment of accident tolerant fuel (ATF) in current PWR designs. The plant model is capable of assessing the safety and performance of different ATF designs in a broad spectrum of anticipated operational occurrences (AOO) and design basis accidents (DBA).

The purpose of accident analysis is to demonstrate compliance of plant performance against applicable regulatory acceptance criteria, thus assuring nuclear safety under postulated off-normal plant conditions. An example of regulatory acceptance criterion is that the calculated maximum temperature of fuel element cladding not be greater than 1200°C (2200°F) for light-water reactors (LWR) fueled with uranium oxide pellets within cylindrical zircaloy cladding. There are generally two broad approaches in performing accident analysis, differentiated by using either conservative or realistic evaluation models. A realistic or best-estimate calculation uses modeling that attempts to describe realistically the physical processes occurring in a nuclear reactor. As a condition for using best-estimate calculations in licensing action, the U.S. Nuclear Regulatory Commission (USNRC) requires the licensee to demonstrate that the code and models used are acceptable and applicable to the specific facility over the intended operating range and must quantify the uncertainty in the specific application. In order to assure compliance with high probability that the calculated results will not exceed the acceptance criteria there is the requirement to consider the uncertainty and quantify it when comparing the results of the realistic or best-estimate evaluation model with the applicable regulatory acceptance criteria or limits. This assurance is further extended by specifying the operation envelope of the nuclear reactor to allow for a safety margin between a limiting safety parameter (e.g. the peak clad temperature (PCT)) and the applicable regulatory limit (e.g. PCT < 1200°C) under the most limiting postulated design basis accident conditions. In the context of our analysis, the safety margin is taken as the difference in physical units between the regulatory acceptance criteria and the results provided by the calculation of the relevant plant parameter after given considerations for conservatism or the uncertainties in the calculations. Figure 1 illustrates the relation among best-estimate result, uncertainty, safety margin, and regulatory limit. Results from best-estimate calculations are used to assess the values of safety margins.

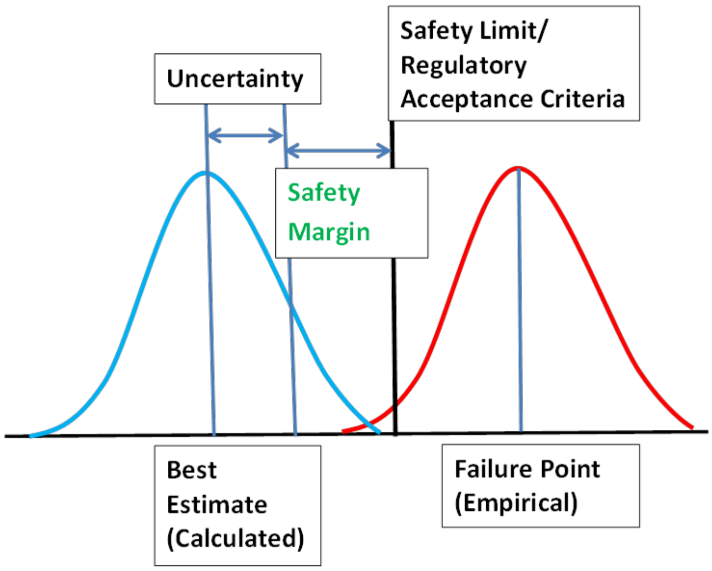

FIG. 1. Safety Margin.

TRACE - **T**RAC/**R**ELAP **A**dvanced **C**omputational **E**ngine [1] is the latest, best-estimate reactor systems code developed by the USNRC for analyzing transient and steady-state neutronic-thermal-hydraulic behavior in LWRs. It combines the capabilities of the USNRC's four main systems codes, TRAC-P, TRAC-B, RELAP5 and RAMONA. A TRACE PWR plant model, which simulates a Westinghouse 2308-MWt three-loop PWR, has been developed for the analysis of AOOs and DBAs. The following is a summary of the model development effort and results of four accidents: loss of off-site power (LOOP), steam generator tube rupture (SGTR), small-break loss-of-coolant-accident (SBLOCA), and large-break loss-of-coolant-accident (LBLOCA).

2. Model Development

The basis for the TRACE model developed for the evaluation of ATF is an example PWR model described in the TRAC-M Users Manual [2]. The TRAC-M example PWR is a Westinghouse 2308-MWt, 3-loop plant. The original model has been transformed to comply with the functionality of the current TRACE input specifications. In particular, all the deprecated models are replaced with their current equivalents and in some cases, components are replaced or renodalized based on the current guidance for the modeling of reactors with TRACE. The PWR plant model, shown in Fig. 2, includes the following salient features:

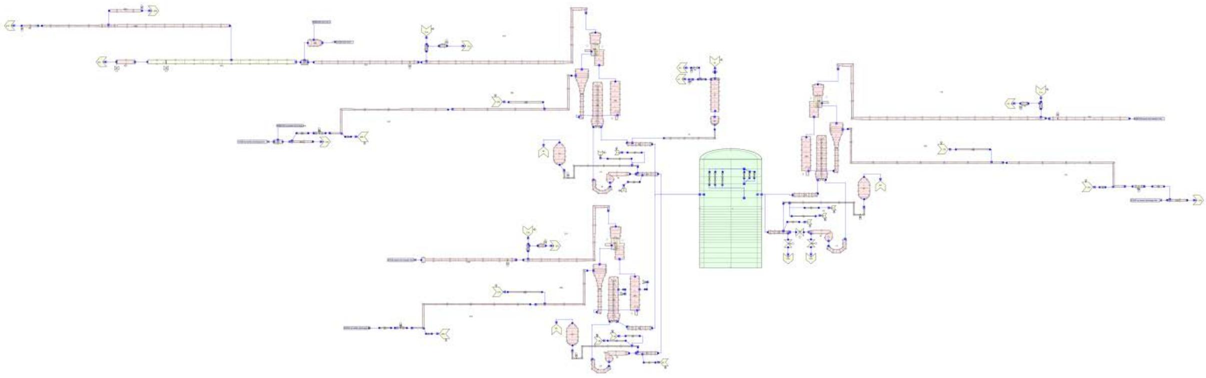

FIG. 2. TRACE PWR Plant Model.

- Three-dimensional reactor vessel (2 radial zones, 6 azimuthal sectors, and 12 axial levels)
- Powered-rod and unpowered-slab heat structures
- Three primary- and secondary-coolant loops modeled individually
- Makeup, letdown, and pressurizer-sprayer chemical and volume control system (CVCS) flows
- Accumulator, low and high pressure safety injection (LPSI and HPSI) fills in each primary-coolant loop
- Pressurizer and pressurizer power-operated relief valve (PORV) and safety relief valve (SRV)
- Main steam and steam-dump lines
- Main and auxiliary feedwater systems
- Various trips and control systems to model plant responses

Figure 3 is a schematic representation of the primary system, showing the three primary loops, the steam generators (SG1 – SG3), the reactor vessel, and heat structures (HS801 – HS806) representing fuel assemblies in the core. The reactor vessel is modeled with two radial rings and six azimuthal sectors. The inner ring represents the core region and the outer ring represents the downcomer of the vessel. As shown in Fig. 3 the hot leg and cold leg of each loop are associated with two adjacent sectors. Each sector has one heat structure representing the fuel assemblies. The current model assumes uniform power for all six fuel assembly heat structures (HS 801 through 806). Each fuel assembly heat structure has two rod types, average and hot. The hot rod has a power factor of 1.678 times the average rods. Figures 4 and 5 present an expanded view of the components modeled in TRACE for the primary loop and secondary loop (main steam and feedwater loop) respectively. In this model, the pressurizer is connected to the hot leg of loop three.

The steady-state operation of the PWR plant is maintained by a number of process control systems. The flow area of the turbine control valve is regulated to maintain the steam generator steam dome to a set pressure. The set pressure corresponds to the condition of the plant that produces the rated steam output, as specified in the plant thermal balance. A three-element control logic is implemented to regulate the feedwater flow to maintain the steam generator water level to a set value. The logic processes the combination of water level error and the steam-feed mismatch to produce a bias adjustment to the feedwater valve flow area demand. The controller output is used to set the feedwater valve flow area.

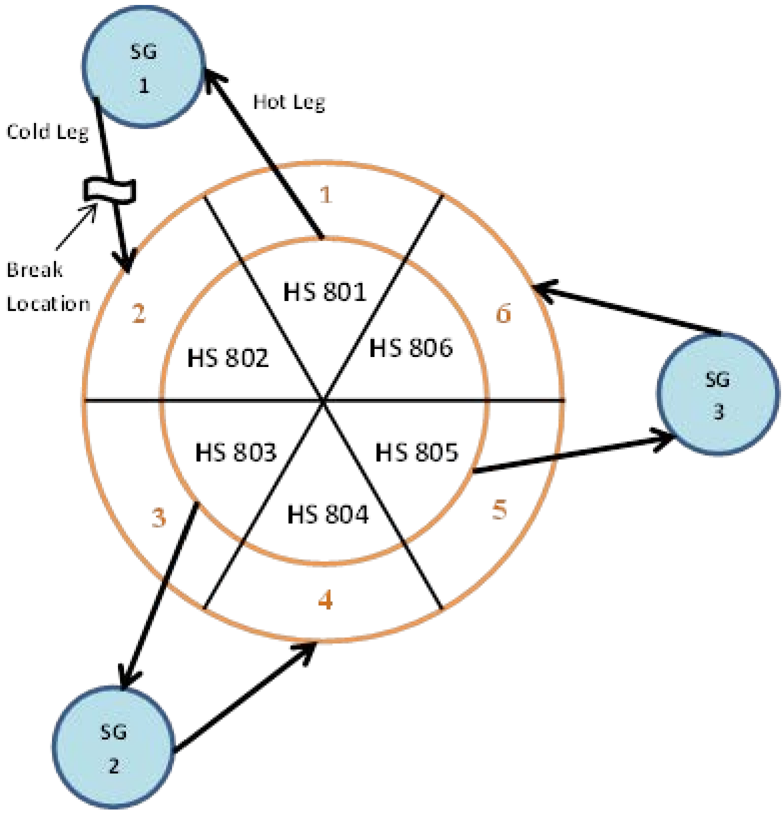

FIG. 3. Three-Loop PWR Model.

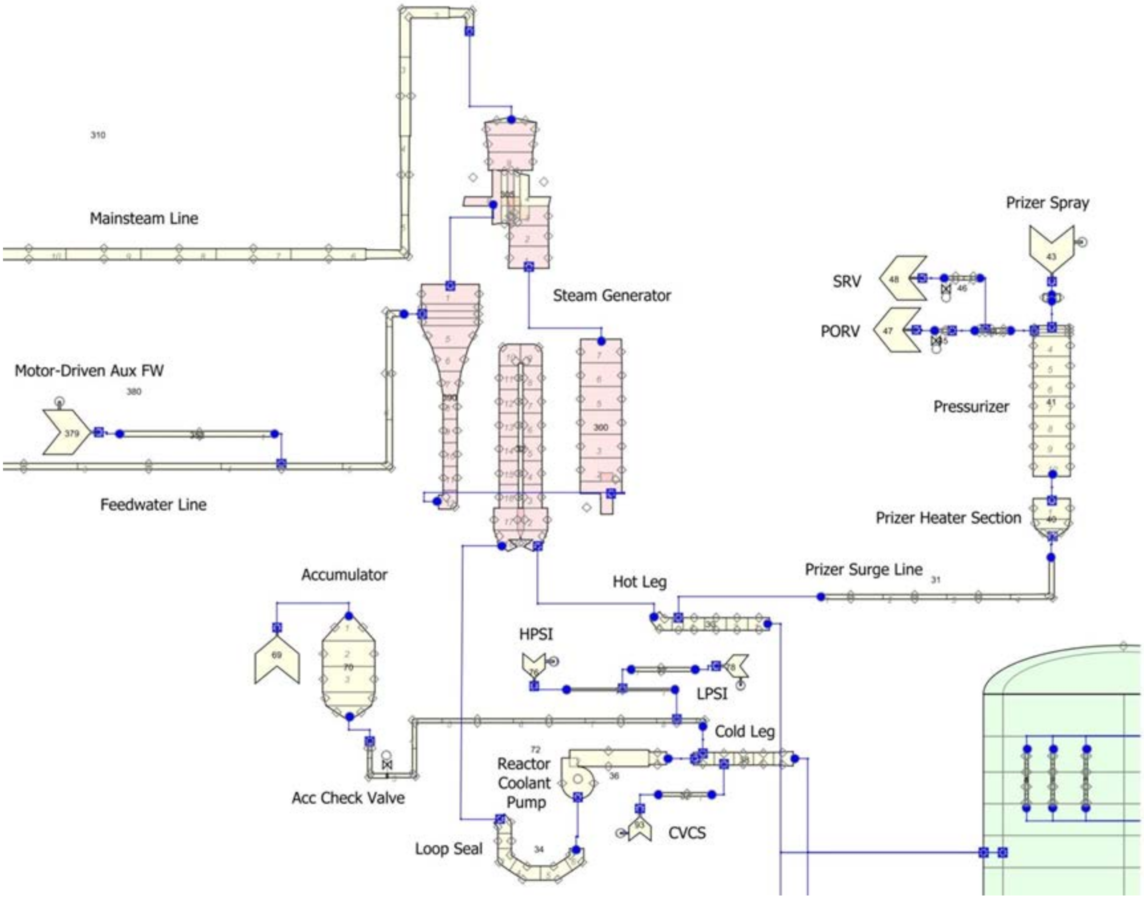

FIG. 4. Primary Loop with Pressurizer.

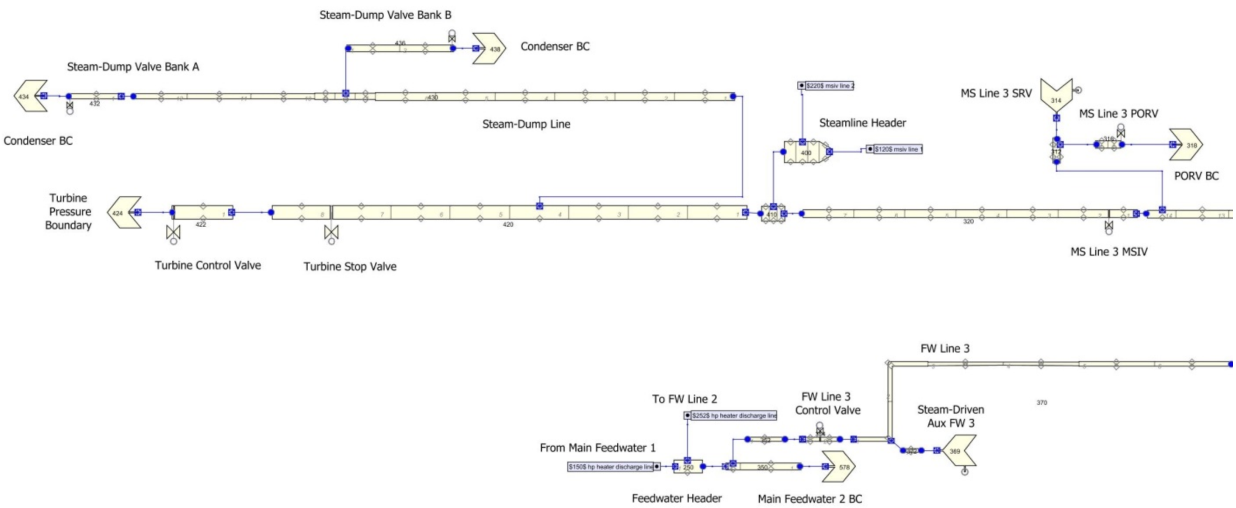

FIG. 5. Main Steam Line and Feedwater Line.

The steady-state of the PWR plant is achieved by using the following tuning strategy:

- A reference plant is used for specifying the core thermal power, primary flow rate and feedwater temperature.
- The desired steam generator pressure is achieved by controlling the flow area of the turbine control valve.
- The desired water level in the steam generator is achieved by a three-element controller (steam flow, feed flow, and water level).
- The desired hot/cold leg temperatures are achieved by adjusting the effective tube-bundle heat transfer area of the steam generator.
- The desired recirculation ratio of the steam generator (ratio of total flow to steam flow) is achieved by adjusting the flow resistance in the steam generator.

3. Kinetics Model

Kinetic parameters for uranium oxide fuel and nitride fuel (UN-U_3Si_2) have been incorporated in the PWR model. Feedback coefficients included in the model are fuel temperature (Doppler), coolant temperature (spectral temperature), coolant density (void), and boron. PARCS [3] stand-alone full-core calculations provide kinetics parameter inputs to the TRACE point-kinetics model. These point kinetic parameters are for beginning-of-cycle conditions for an equilibrium cycle model based on a reference PWR. It is significant to note that the delayed neutron fractions (β) and other kinetics parameters of the two cores are different (for example, β_{UO2} = 0.00598 and $\beta_{UN\text{-}U3SI2}$ = 0.00612). In the PARCS core calculations, the few-group parameters were developed utilizing the TRITON/NEWT tools in the SCALE package [4]. A simplified set of branch cases was included in the few-group parameters. Separate sets of parameters (prompt-neutron lifetime and group constants for six delayed-neutron groups) have been calculated for the standard UO_2 fuel and the UN-U_3Si_2 fuel. In addition, SCRAM reactivity curves and reactivity coefficients are determined from PARCS kinetics calculations, assuming all control rods are Ag-In-Cd type. Table 1 shows the four reactivity coefficients (fuel temperature (T_f), moderator temperature (T_m), moderator density (D_m), and boron).

Table 1. FULL-CORE REACTIVITY COEFFICIENTS FOR CANDIDATE FUELS

	UO_2	$UN\text{-}U_3Si_2$
T_f (pcm/K)	-2.74	-2.83
T_m (/K)	1.93	0.862
D_m (pcm/kg/m^3)	14.2	14.0
Boron (pcm/ppm)	-5.99	-4.49

4. Accident Analysis

Four accidents, one AOO and three DBAs, have been simulated to demonstrate the applicability of the TRACE PWR model to assess the safety performance of ATF. Zircaloy clad UO_2 fuel and nitride fuel ($UN\text{-}U_3Si_2\text{-}UB_4$) are considered. The four accidents are: loss of off-site power (LOOP), steam generator tube rupture (SGTR), small-break loss-of-coolant-accident (SBLOCA), and large-break loss-of-coolant-accident (LBLOCA).

4.1. Loss of Off-Site Power

The loss of off-site power (LOOP) is an AOO. With the loss of electrical power, a primary pump trip at time zero initiated the transient and a reactor trip on low flow with a one-second time delay occurred at 3 s. Reactor power and primary flow are shown in Figs. 6 and 7 respectively. The decrease in reactor power before the scram is due to a negative reactivity feedback from the coolant heat-up (decrease in coolant density) because the pump trip reduced the primary flow. Figure 8 illustrates the fuel temperature response in an average fuel rod. The average fuel temperature is for an axial node near the core mid-plane (each fuel rod has sixteen uniform axial nodes). The UN fuel exhibits a lower average temperature than

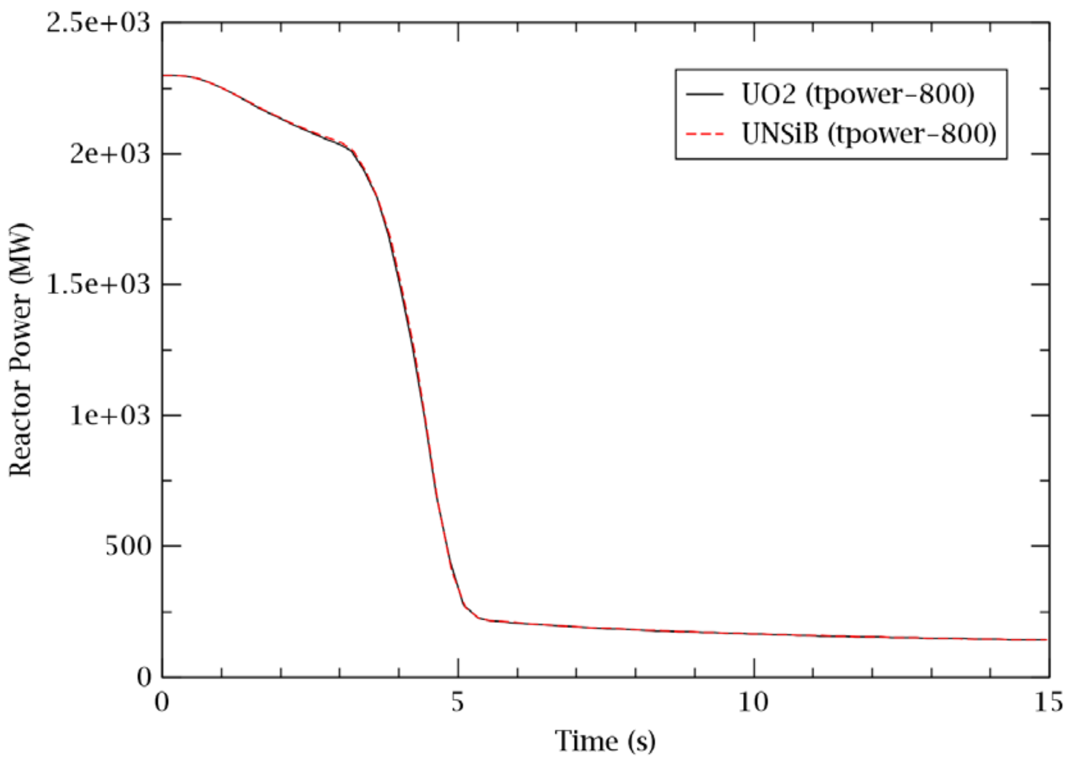

FIG. 6. Loss of Off-Site Power – Reactor Power.

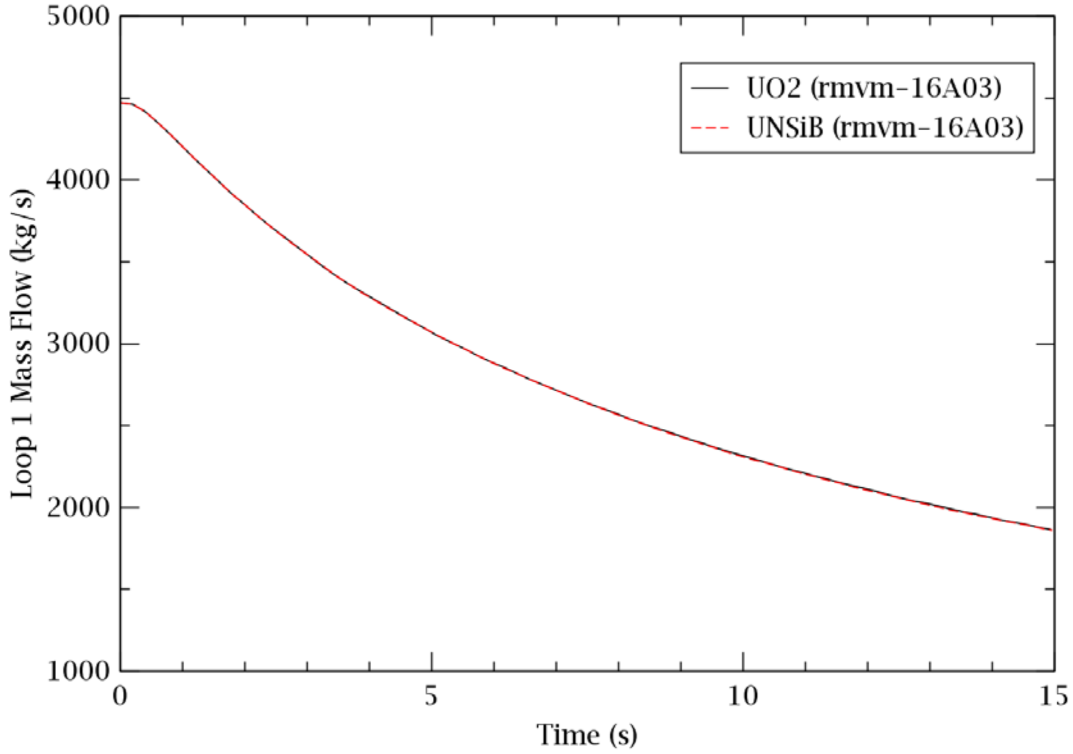

FIG. 7. Loss of Off-Site Power – Loop 1 Mass Flow.

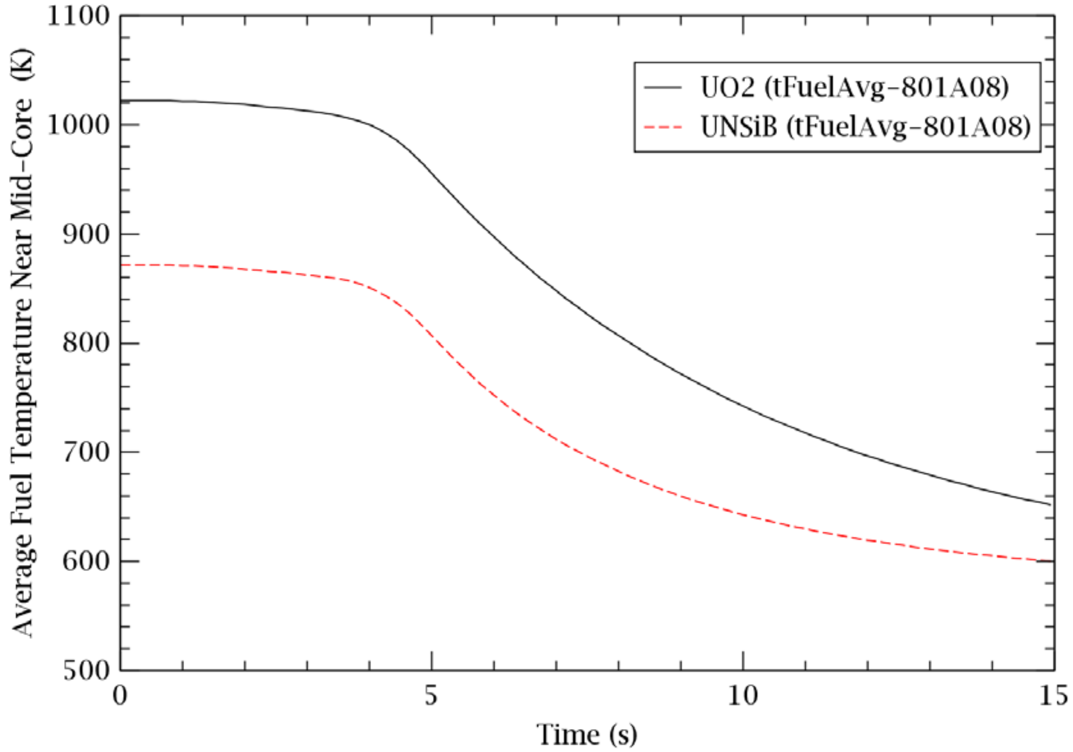

FIG. 8. Loss of Off-Site Power – Average Fuel Temperature in Axial Node 8.

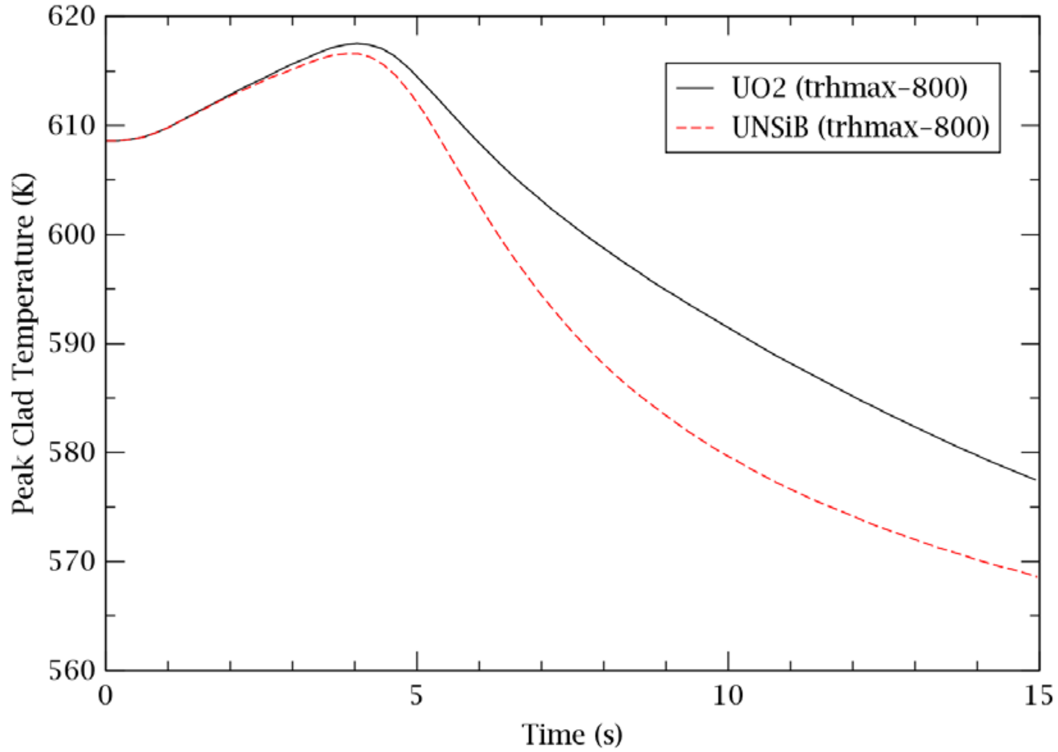

FIG. 9. Loss of Off-Site Power – Peak Clad Temperature.

the UO_2 fuel because it has a higher thermal conductivity. A lower fuel temperature in the UN fuel also implies it has a lower stored energy than the UO_2 fuel. It is noted that the UN-U_3Si_2-UB_4 fuel and the UO_2 fuel have similar volumetric heat capacity at temperatures near the normal operating range. A lower initial temperature and a lower stored energy cause the UN fuel to have a lower fuel and clad temperatures than the UO_2 fuel after the reactor scram. The PCT for the two fuel types is compared in Fig. 9. Overall, the fuel response in an LOOP is relatively mild because the power decay is faster than the flow decay. The difference in the fuel temperature for the two fuel types is a reflection of the reduced stored energy and higher thermal conductivity for the nitride fuel.

4.2. Steam Generator Tube Rupture

A steam generator tube rupture (SGTR) accident has been analyzed using the TRACE PWR model with point-kinetics. The single-tube rupture was simulated by a restart run in which a pipe was added to the model connecting the primary and secondary sides of the loop 2 steam generator. The location of the rupture connection is shown in Fig. 10. In the restart run, the new PIPE component provided a rupture flow area (6.087×10^{-4} m^2) equal to twice the flow area of a single tube, representing a double-ended guillotine break. In an SGTR transient, the reactor loses pressure due to leakage of the primary coolant to the secondary side of the steam generator.

The SGTR is a rather slow transient and the TRACE results have been examined to confirm the operation of various trips and controls. A comparison of the output against the input model verifies the sequence of trips generated by the SGTR transient. These trips include: reactor trip on low pressurizer pressure, turbine trip on reactor trip, initiation of safety injection (SI) on low pressurizer pressure, main feedwater pump trip on SI signal,

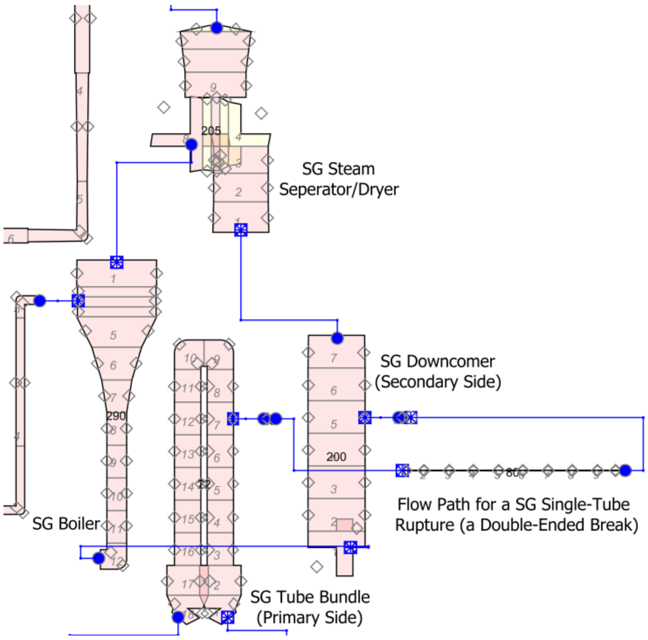

FIG. 10. Break Connections for Steam Generator Tube Rupture.

initiation of motor-driven and steam-driven auxiliary feedwater pumps, and primary pump trip on low-low pressurizer pressure.

Figure 11 compares the PCT for the oxide fuel and the nitride fuel. The PCT generally reflects the reactor power and core flow. It drops rapidly after the reactor trip at ~180 s and slowly increases after the primary pump trip at ~210 s. The reactor power, depicted in Fig. 12,

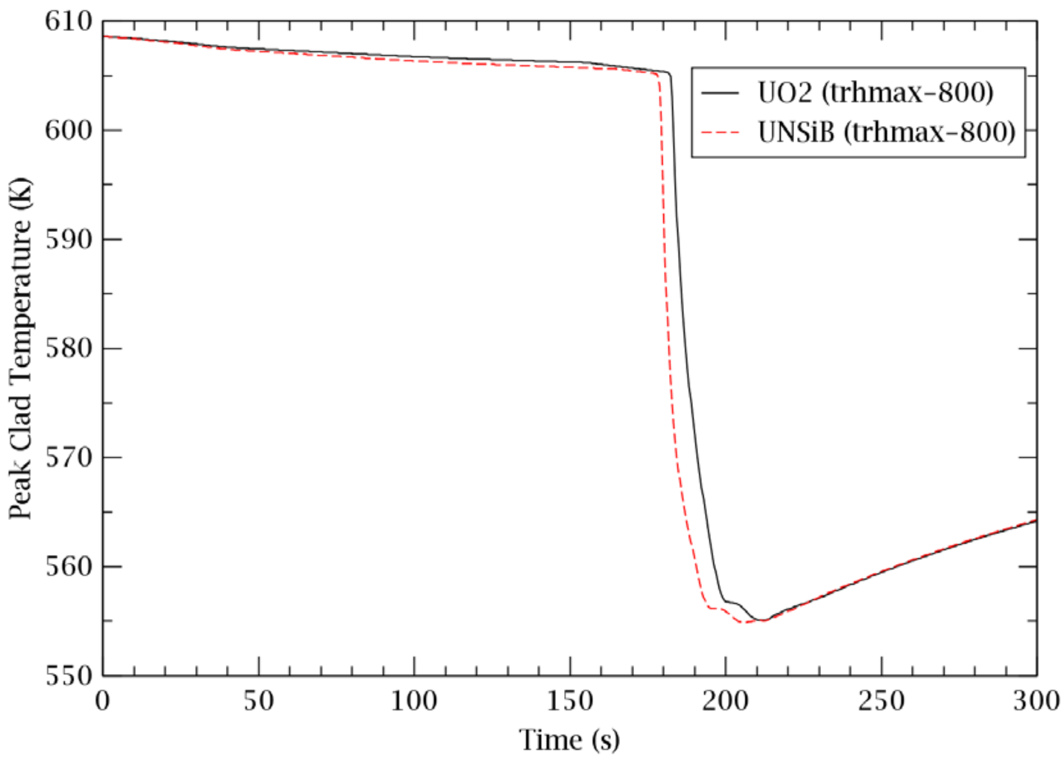

FIG. 11. Steam Generator Tube Rupture – Peak Clad Temperature.

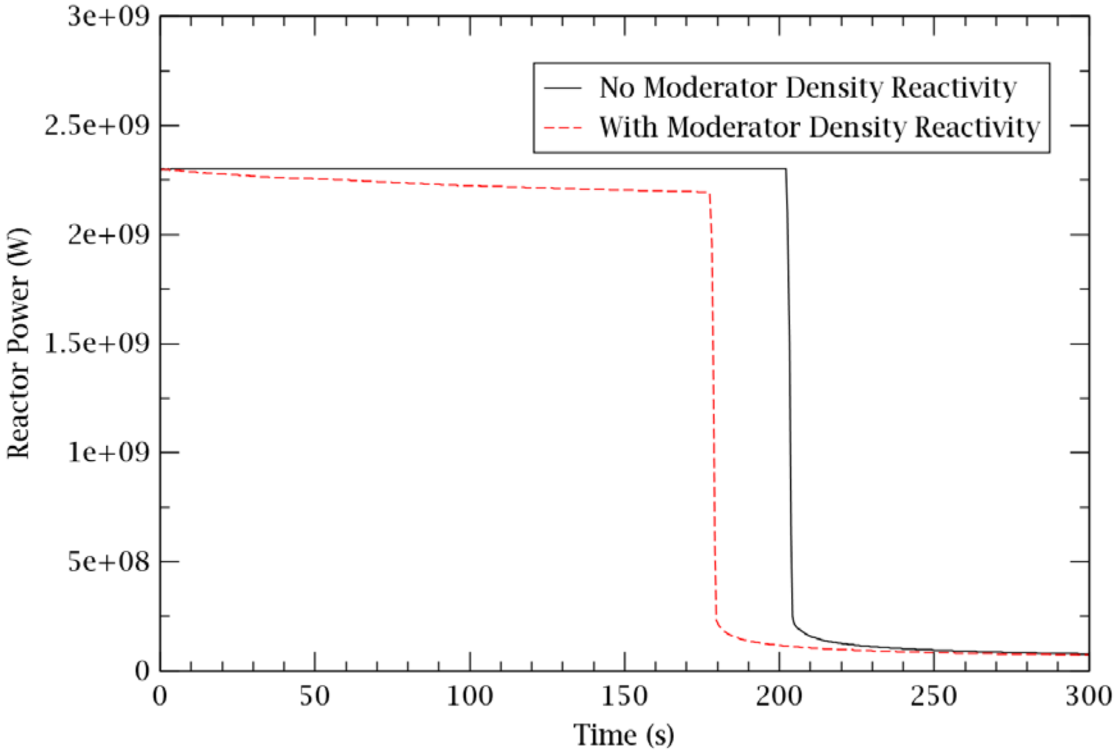

FIG. 12. Steam Generator Tube Rupture – Reactor Power.

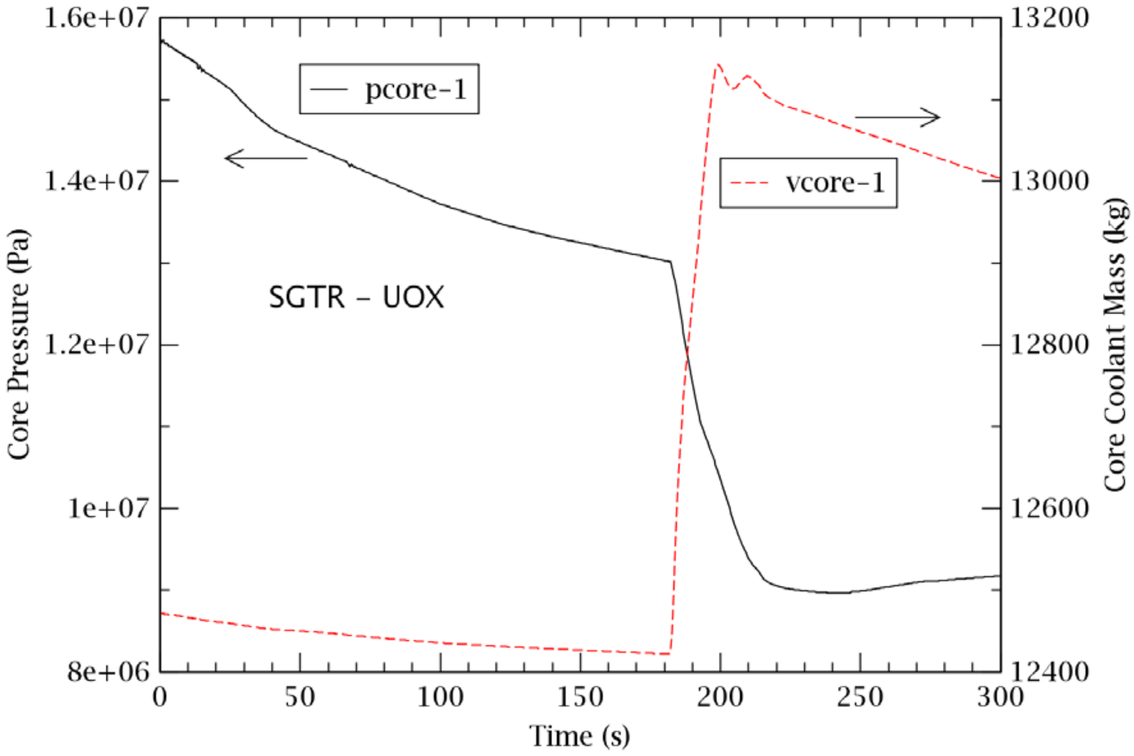

FIG. 13. Steam Generator Tube Rupture – Core Pressure and Core Coolant Mass.

shows a slow decline until the scram when the power drops rapidly. Reactivity feedback from decreasing coolant density is the cause of the small decrease in reactor power before reactor trip. This is confirmed by a test case in which the moderator density reactivity was turned off. Result of the test case (no moderator density feedback) in Fig. 12 shows a steady reactor power until scram. The decrease in moderator density can be inferred from the core coolant mass in Fig. 13 (for the nitride fuel case). Before reactor scram, the coolant mass decreases with the decreasing reactor pressure (a lower density for lower pressure). After the reactor scram, steam generators continue to remove heat from the reactor and this leads to a decrease in the coolant temperature and an increase in the core coolant mass (a higher density for lower temperature). A subsequent decrease in the core coolant mass occurs after the primary pump trip that reduces both the primary flow and the heat removal by the steam generators.

4.3. Small-Break LOCA

A small break in the cold leg of primary loop 1 was simulated with a loss of off-site power coincident with the reactor trip. The loss of off-site power assumption led to the tripping of primary pumps, main feedwater, and turbine stop valve. In the SBLOCA transient, the pipe rupture was initiated by opening a break valve (Valve 704 in Fig. 14) at time zero simulating a 0.1 m^2 break that opened completely in 0.001 s. This particular transient exercised a broad spectrum of systems in the model, including primary, secondary, and ECCS (emergency core cooling system).

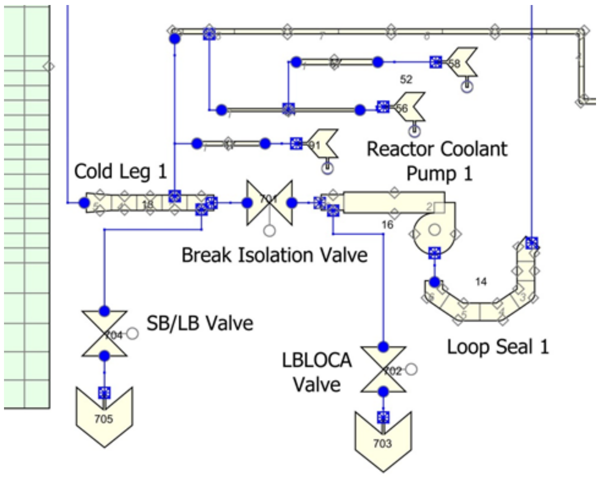

FIG. 14. Break Connection for Pipe Rupture and Double-Ended Break.

The SBLOCA analysis considered two types of fuel, the reference uranium oxide fuel (UO_2) and a nitride fuel (UN-U_3Si_2-UB_4). The progression of the SBLOCA for the two types of fuel is generally quite similar. Any variations are mainly due to thermal-physical property differences and thermal-hydraulic response of the system. Figure 15 shows the break flow and the total emergency core cooling system (ECCS) flow for this SBLOCA. The two flows are similar for the two fuel types in general. There are changes in the slope of the break flow at ~10 s and 45 s. The slowing down of the break flow at ~10 is due to the beginning of voiding in the cold leg in loop 1, i.e. the break flow changes from single-phase liquid to two-phase mixture. At ~ 45 s, the break flow increases again because of reduction in voiding in the cold leg of loop 1 due to significant injection of ECCS flow. In a SBLOCA, the break flow rate generally decreases with increase in voiding because of the choked flow condition at the break. High pressure safety injection (HPSI) was initiated at ~23 s but the ECCS flow was insignificant until the accumulator flow started at ~44 s. The reactor pressure in this analyzed accident did not decrease low enough for the low pressure safety system to begin injection.

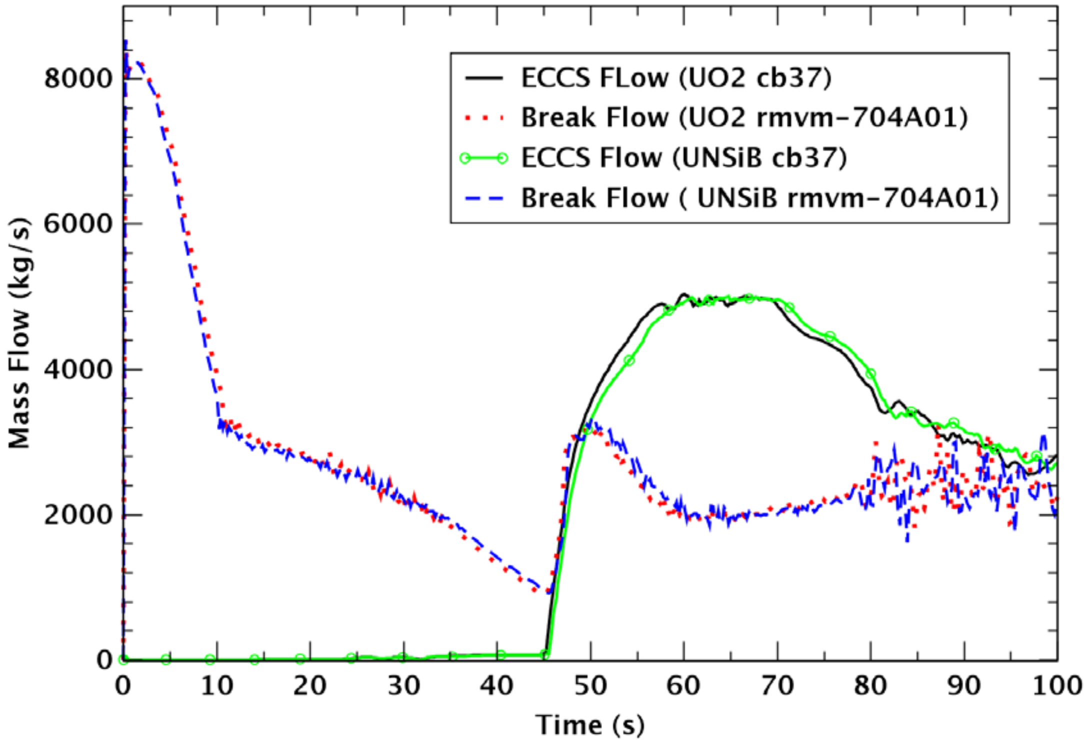

FIG. 15. SBLOCA – Break Flow and ECCS Flow.

Figure 16 shows the peak clad temperature (PCT) for the two fuel types. For the oxide fuel, the PCT occurs in the hot rod in sector 2 (heat structure 802001). For the nitride fuel, it is the hot rod in sector 6 (heat structure 806001) that exhibits the PCT. In the SBLOCA, the nitride fuel shows a higher PCT than the oxide fuel. The magnitude of the PCT can be correlated with the duration of the state in which the fuel rod is uncovered i.e., not cooled by coolant. Figure 17 shows the location of the lower quench front for the two hot rods corresponding to the UO_2 fuel and the nitride fuel respectively. In Fig. 17, a location of 4.27 m corresponds to the top of the core and the zero corresponds to the bottom of the core. Heatup of the fuel, as seen in the rise in the PCT, begins when the quench front disappears to the bottom of the core and the subsequent recovery of the quench front brings a corresponding drop in the PCT. It is also informative to review the corresponding average fuel temperatures shown in Fig. 18. The nitride fuel with a higher thermal conductivity than the oxide fuel initially exhibits a lower temperature. By about 40 s, the two fuel types have reached about the same average fuel temperature. From that point on the PCT for the two fuel types begins to differ and the longer the fuel remained uncovered the higher the PCT becomes. Since the bottom quench front for the nitride fuel stayed below the core inlet for a longer period, its PCT reaches a higher value than the UO_2 fuel.

Results of the SBLOCA suggest that global system response alone does not provide sufficient information to delineate the fuel performance under accident conditions. In addition, the accident simulation model should have sufficient fidelity to provide local core conditions As seen in this SBLOCA, though the break flow and the ECCS flow are quite similar for the two fuel types, differences in the fuel response are driven by complex core thermal-hydraulics.

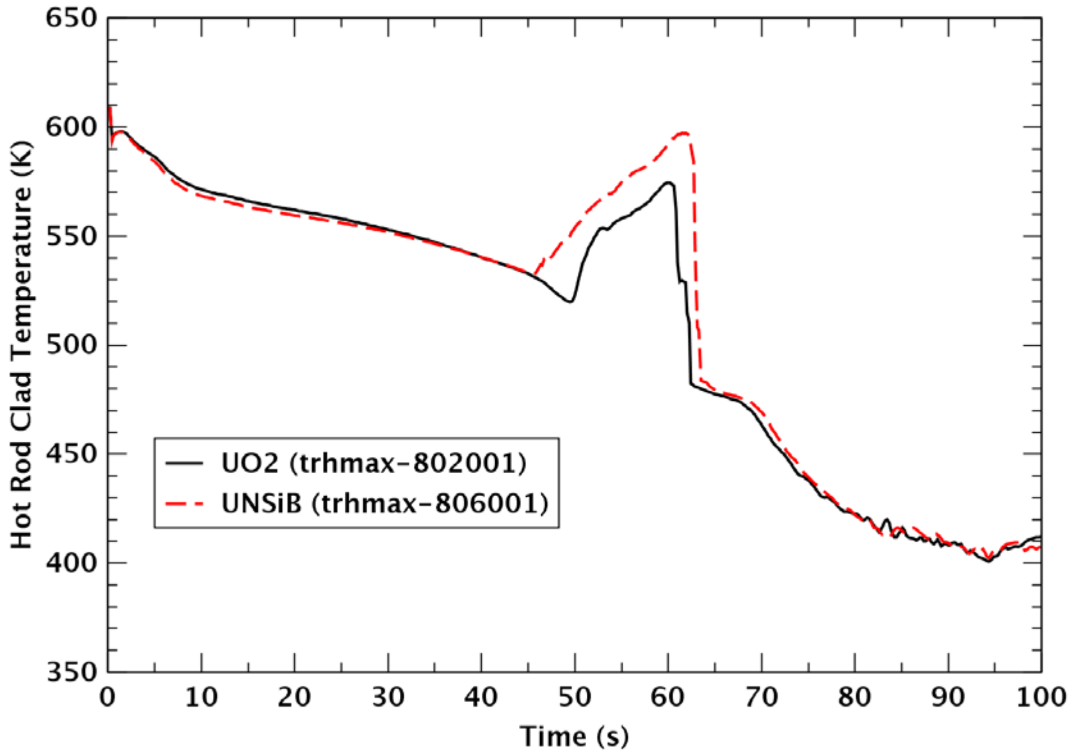

FIG. 16. Small-Break LOCA – Peak Clad Temperature.

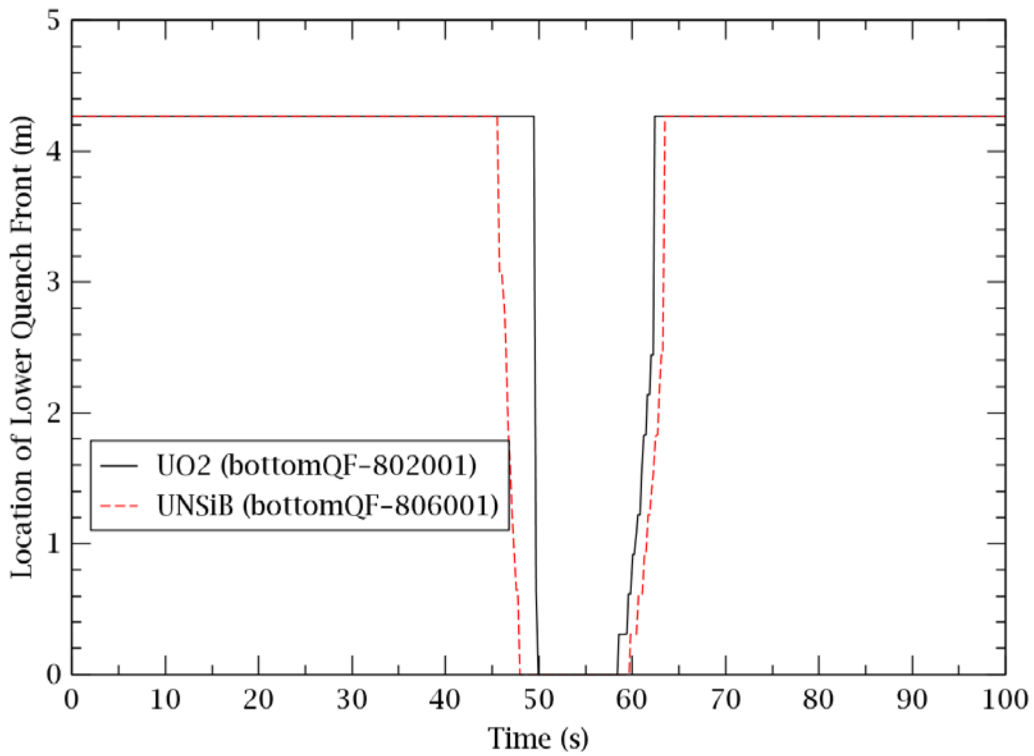

FIG. 17. Small-Break LOCA – Location of Lower Quench Front.

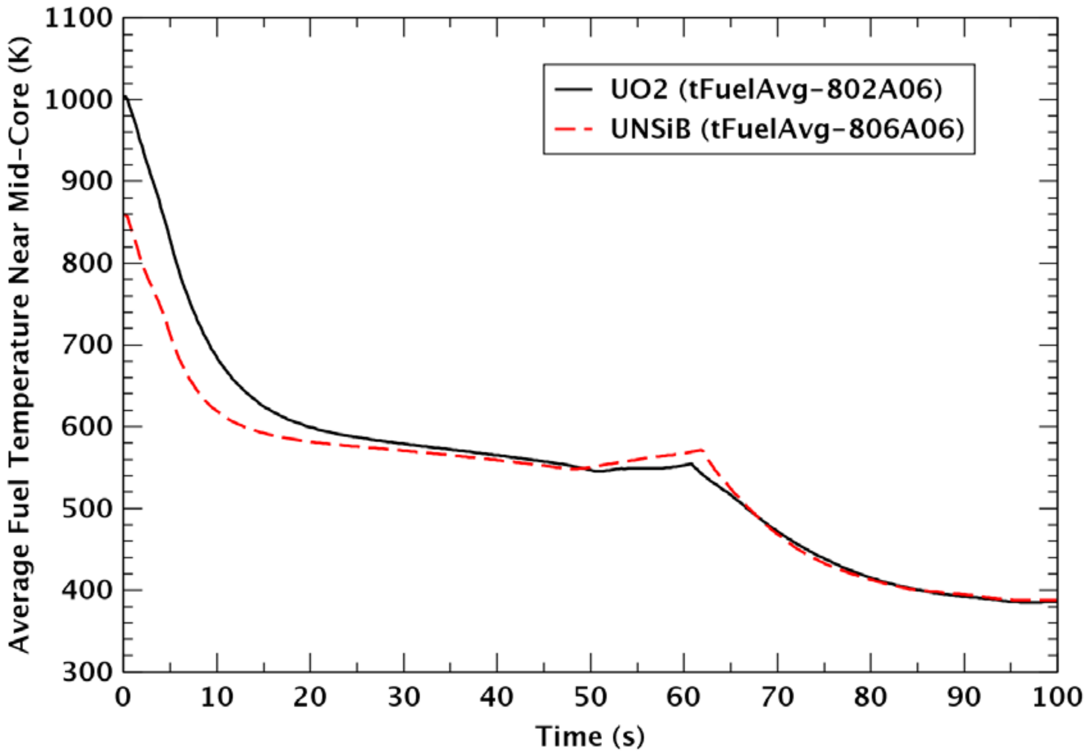

FIG. 18. Small-Break LOCA – Average Fuel Temperature Near Mid-Core.

4.4. Large-Break LOCA

A large break in the cold leg of primary loop 1 was simulated with a loss of off-site power coincident with the reactor trip. In the LBLOCA transient the double-ended pipe break was initiated at time zero by opening the two break valves in Fig. 14 (Valves 702 and 704) and closing the break isolation valve (Valve 701 in Fig. 14). For this fast transient the thermal-hydraulics and fuel thermal properties dictate the overall response more so than the reactivity feedback.

Figure 19 shows the PCT for the two fuel types. The PCT for the nitride fuel is ~30 K lower than the oxide fuel. Figure 20 shows the core liquid fraction. The initial blowdown causes the core to be uncovered with the core liquid fraction dropping to zero at about 20 sec. This leads to fuel heatup and a higher clad temperature. With the initiation of ECCS flow (see Fig. 21), the core liquid fraction begins to increase at ~35 s. Refilling and reflooding of the core quench the fuel rods with a consequential reduction in the clad temperature. The ECCS flow starts with flow from the accumulator followed by initiation of the high pressure and low pressure injection systems. In the LBLOCA simulation, the ECCS flow in loop 1 was disabled because the flow was assumed to leak out of the break and never reached the core.

It is of interest to note the asymmetry in the system response. Figure 22 shows the PCT for the hot rods in sectors 1 and 2. According to the loop layout in Fig. 3, all break flow has to pass through sector 1 first. Thus, sector 1 experiences the highest flow among the 6 sectors. As a result, the fuel rods in sector 1 receive the most cooling and register the lowest PCT among fuel assemblies in all 6 sectors.

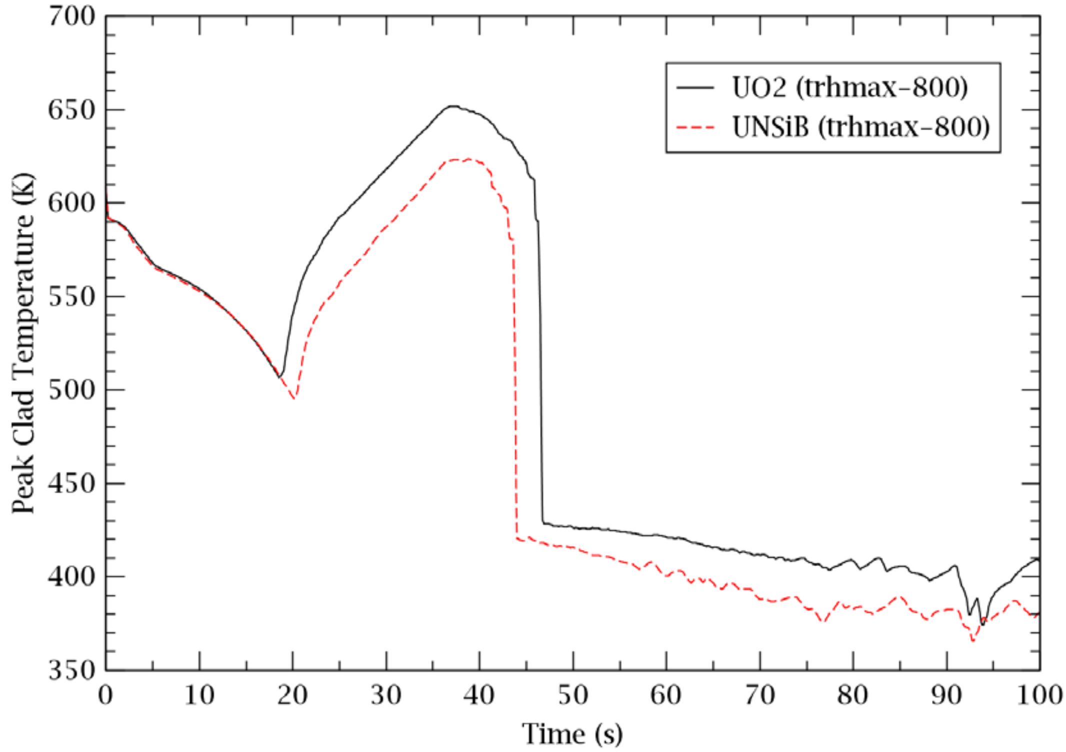

FIG. 19. Large-Break LOCA – Peal Clad Temperature.

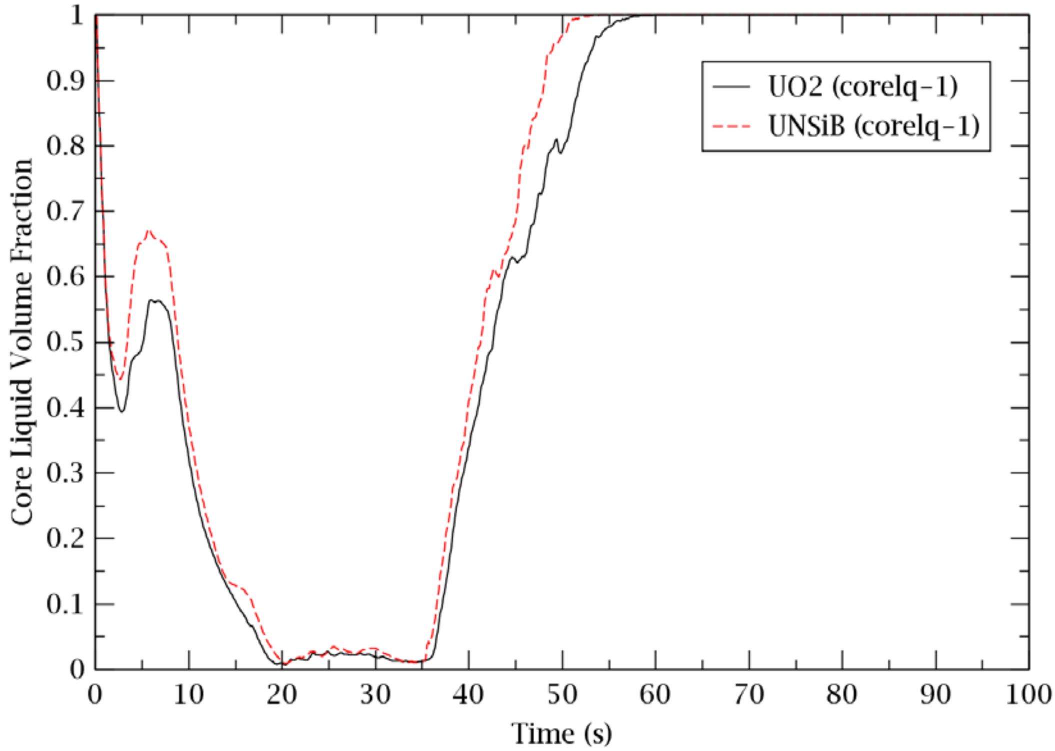

FIG. 20. Large-Break LOCA – Core Liquid Fraction.

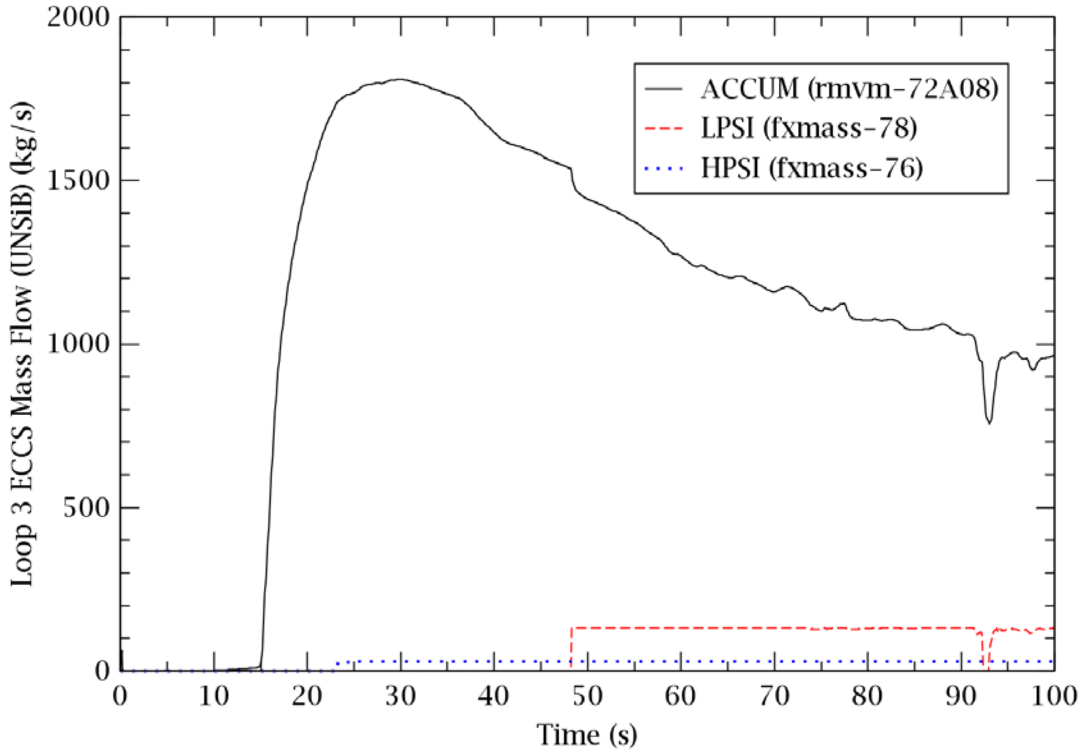

FIG. 21. Large-Break LOCA – Loop 3 ECCS Flow.

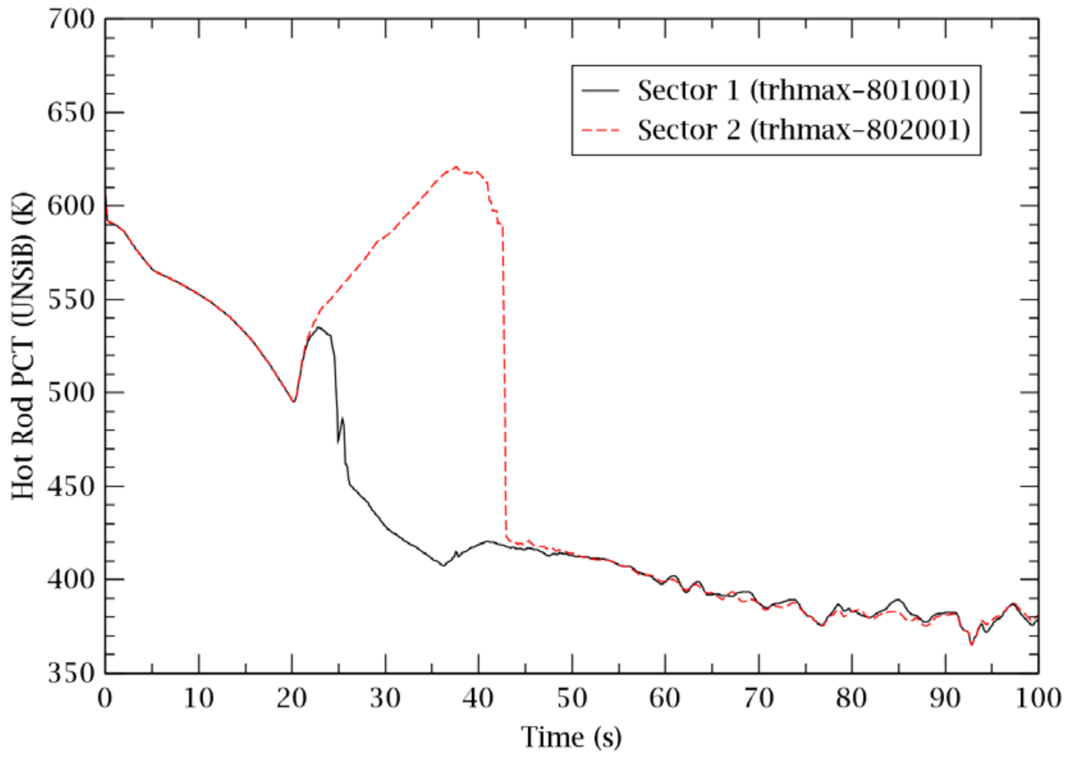

FIG. 22. Large-Break LOCA – PCT in Sectors 1 and 2.

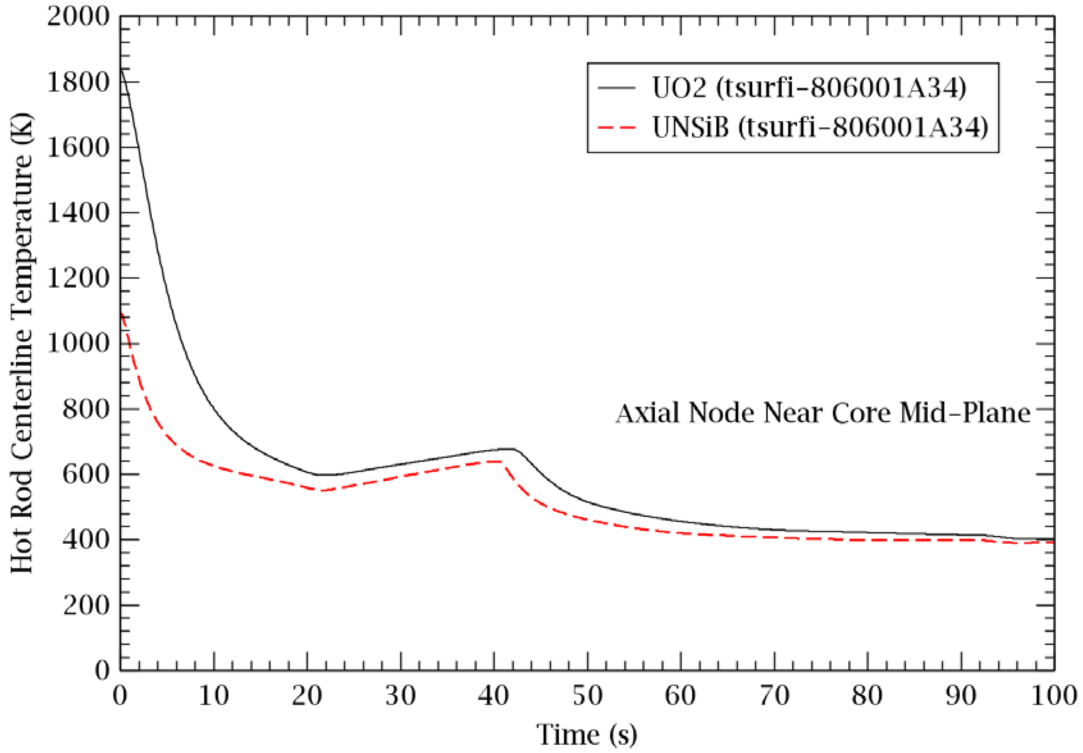

FIG. 23. Large-Break LOCA – Fuel Centerline Temperature.

The fuel centerline temperature for the hot rod near the core mid-plane is shown in Fig. 23. In general, the nitride fuel is at a lower temperature than the oxide fuel because it has a higher thermal conductivity than the UO_2 fuel.

In addition to calculating the transient fuel and clad temperatures, this simulation also informs on the axial temperature distribution. Figures 24 and 25 show the axial distribution of the centerline temperature and the clad temperature respectively for the oxide fuel hot rod at various times during the LBLOCA. The heatup and quenching of the fuel are readily seen in their manifestation in the axial temperature distribution. In Fig. 25, the effect of top reflood is seen by comparing the clad temperature near the top of the core at 20 s and 30 s. The effect of bottom refill is seen by comparing the clad temperature in Fig. 25 near the bottom of the core at 30 s and 40 s.

The LBLOCA results again accentuate the influence of core thermal-hydraulics on the fuel response in an accident. The asymmetric core condition is seen to result in very different PCT. Core uncovery translates to core heatup. On the other hand, core refill and reflood provide the needed cooling to limit the PCT. Contrary to the result of the SBLOCA, in the case of the LBLOCA the period of core uncovery is shorter for the nitride fuel than the oxide fuel resulting in a PCT that is 30 K lower for the nitride fuel than the oxide fuel.

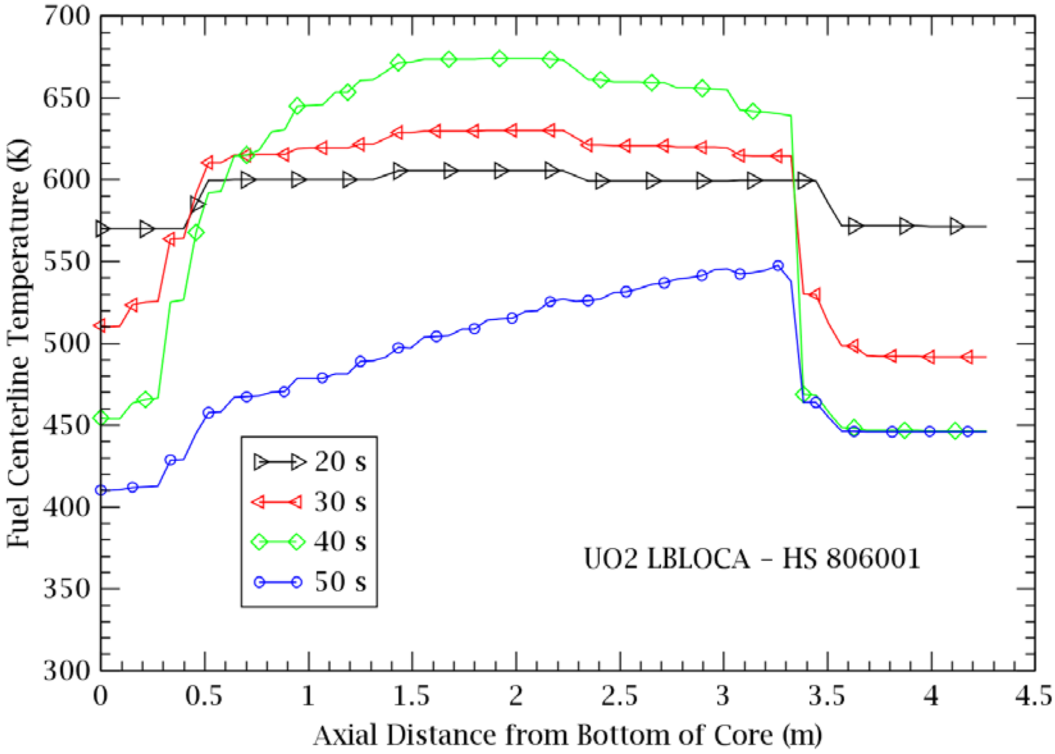

FIG. 24. Large-Break LOCA – Axial Distribution of Fuel Centerline Temperature.

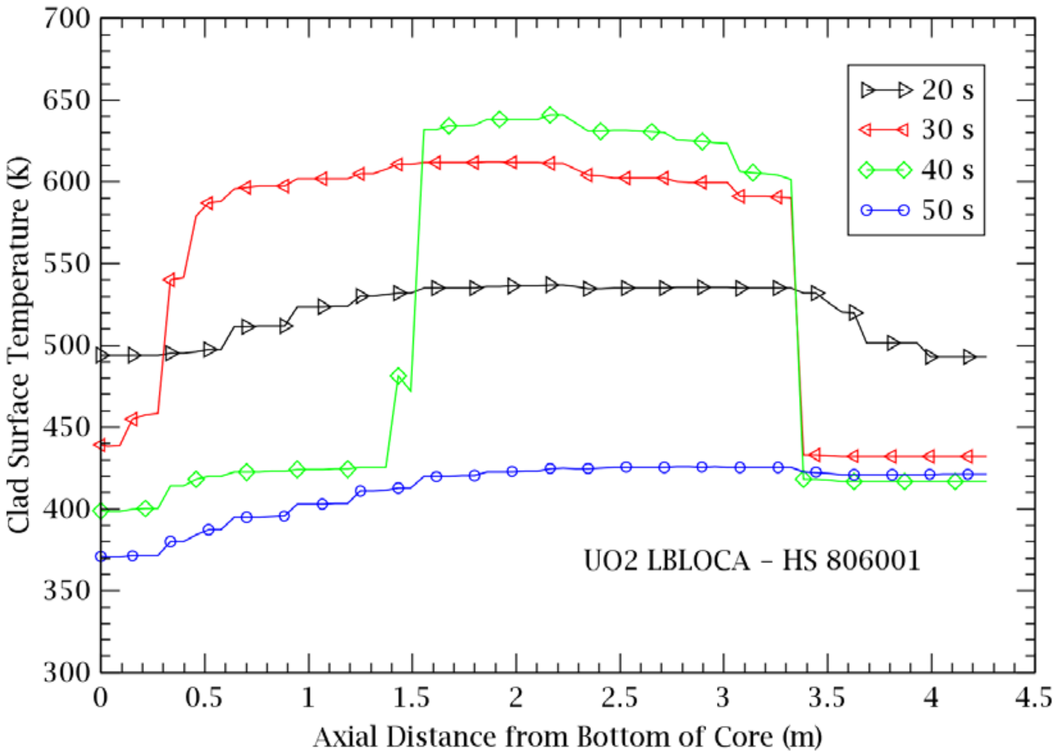

FIG. 25. Large-Break LOCA – Axial Distribution of Clad Surface Temperature.

5. Summary and Conclusion

A TRACE PWR plant model has been developed and applied to analyze different accidents and transients, including design basis accidents and anticipated operational occurrences. Its capability has been demonstrated and its fidelity is being improved with continuing development. This model is capable of providing transient and spatial temperature information for the fuel and cladding. This simulation platform supports the development of advanced fuels and claddings by providing a scoping capability to evaluate their impact on reactor performance and safety.

The transient and accident events show a very similar response for UO_2-Zr fuel-clad and UN-U_3Si_2-Zr fuel-clad. In the LBLOCA, LOOP, and SGTR the improved thermal properties of UN-U_3Si_2 result in enhanced thermal margin, a lower peak clad temperature than the oxide fuel. Among the accidents analyzed, the largest performance difference between the oxide fuel and the nitride fuel is in the LBLOCA where the peak clad temperature (PCT) for the nitride fuel is 30 K lower than that of the oxide fuel. In the SBLOCA, the period of uncovered core lasted a little longer for the nitride fuel resulting in a higher PCT than the oxide fuel. Both the SBLOCA and the LBLOCA analysis accentuate the influence of core thermal-hydraulics on the fuel response in an accident. Among the four accidents analyzed, only the slower transients with delayed reactor scrams, namely the LOOP and the SGTR, do the effects of reactivity feedback become discernible.

ACKNOWLEDGEMENTS

This manuscript has been authored by employees of Brookhaven Science Associates, LLC under Contract No. DE-AC02-98CH10886 with the U.S. Department of Energy. The publisher by accepting the manuscript for publication acknowledges that the United States Government retains a non-exclusive, paid-up, irrevocable, world-wide license to publish or reproduce the published form of this manuscript, or allow others to do so, for United States Government purposes.

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